Research Journal of Applied Sciences, Engineering and Technology 3(8): 737-745, 2011
ISSN:2040-7467

Received: May 09, 2011 Accepted: June 15, 2011 Published: August 30, 2011

Analysis of thermal-hydraulic transients for the Miniature Neutron Source Reactor (MNSR) in Ghana

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Abstract: A mathematical model is presented that permits to simulate the effect of the cooling coils of the pool upper section on the reactor thermal-hydraulic behaviour of Ghana research reactor-1. The model is based on a lumped parameter description solved numerically using Matlab/Simulink tool which is a commercial software package with the capability of modelling dynamical and control systems. The model incorporates fuel grids and cooling coil models as well as radiating energy from the clad. In this model, the reactor tank and the pool is divided into three sections. The model predictions are qualified by comparing the results with experimental data. The effect of cooling the upper section of the pool on reactor thermal-hydraulic parameters using the cooling coil is presented and discussed. It was observed that all maximum values of the reactor thermal-hydraulic parameters decrease when the cooling coil power is increased. Good agreement is found between the model predictions and the experimental results.

Key words: Cooling coil, coolant, lumped parameter, natural convection, thermo-hydraulic, tank-in-pool

INTRODUCTION

Ghana Research Reactor-1 (GHARR-1) is a low power Miniature Neutron Source Reactor (MNSR) similar to the Canadian SLOWPOKE reactor. The thermal and hydrodynamic objective of the design is to safely remove the heat generated in the fuel without producing excessive fuel temperatures or steam void formations and without closely approaching the hydrodynamic critical heat flux under steady-state operating conditions. Under normal reactor operating conditions, it is expected that the rate of heat generation in the fuel will be the same as the rate of heat removal by the coolant. Any imbalance in this state is likely to bring about a perturbation, which can lead to an accident.

Analysis of transient behaviour of research reactors is very important because of its relevance in the determination of the limits imposed by clad melting temperature (Housiadas, 2000). The reactor GHARR-1 is a 30 kW tank-in-pool research reactor, which uses 90.2% enriched uranium-Al alloy as fuel.

The reactor is designed to be compact and safe and it is used mainly for neutron activation analysis, production of short-lived radioisotopes and for education and training. The maximum thermal neutron flux at its inner irradiation site is 1×10^{12} n/cm^{2}s (Akaho *et al*., 2003). A detailed description of the operating characteristics of the facility is published elsewhere (Akaho *et al*., 2000; Ahmed *et al*., 2006). The reactor is cooled and moderated with light water, and light water and beryllium act as reflectors. The core has a central guide tube through which a Cd control rod cladded in stainless steel moves to cover the active length of 230 mm of the core. The single control rod is used for regulation of power, compensation of reactivity and for reactor shutdown during normal and abnormal operations (Qazi *et al*., 1996). The reactor core is located in the lower section of a sealed aluminium alloy vessel of diameter 0.6 m which is hanged on a frame across a stainless steel lined water pool of diameter 2.7 m and depth 6.5 m as shown in Fig. 1. The core is cooled by natural convection and under normal operational conditions, the flow regime is single phase but nucleate boiling is expected under abnormal condition when power excursion occurs due to large reactivity insertions.

Thermocouples and level gauges are used for measuring the reactor thermo-hydraulic parameters during reactor operations. These devices monitor the operating conditions of the reactor and provide information through the instruments mounted on the main control console or the computer closed loop system. Measurement of the temperature difference between core inlet and outlet is accomplished with 2 alumel-chromel thermocouples. One of them is located outside the side annular reflector near the core inlet orifice for the measurement of the inlet temperature. The other is at the upper part of the annular reflector near the core outlet orifice to measure the outlet

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temperature. MNSR thermal-hydraulics have been studied (Zhang, 1993), but the study considered a simplified model that it cannot take into account the eventuality of cooling some parts of the tank or the reactor pool by cooling coils. Another important feature for thermo-hydraulic studies of MNSR, intended as with no reactivity insertion, is the structural design of the reactor which differs from one MNSR to another. Again the behaviour of the power up-rise to its operational level also depends on the reactor.

The mechanism of the cooling coil in core heat removal is that water in the cooling coil circulates to a refrigeration unit (chiller) where it is cooled before returning to the coil in the reactor vessel. The cool water in the coil in turn cools the reactor vessel water, which circulates through the reactor core. In addition, heat from the reactor vessel water can transfer through the vessel wall to the reactor pool. The present study aims at studying the effect of the cooling coils of the pool upper section on the reactor thermal-hydraulic parameters by modifying the model described in (Zhang, 1993), in such a way that it can take into account the presence of cooling coils to cool all parts of the pool and the upper section of the tank.

THE SIMPLIFIED THERMO-HYDRAULIC PROBLEM OF GHARR-1

The thermal-hydraulic design of GHARR-1, intended as with no reactivity insertion, is closely dependent on the structural design of the reactor. The core is cooled by natural convection which is established through the heat generated by fission occurring in the core. The reactor coolant is drawn through the inlet orifice by natural convection flow through the channels within the fuel elements.

The coolant moves up through the core (with average temperatureT_{av}) and exits through the core outlet orifice at temperatureT_2 to the upper part of the tank where the temperature isT_3. The coolant exiting the core is supposed to keep the temperatureT_2 up to the level H_1 after which the coolant mixes with water present in the upper part of the tank.

The coolant inside the core passes through the aperture surrounding the upper ends of the fuel. New colder coolants is substituting the hotter one in the core causing the coolant in the downcomer (with temperature T_1) to move downwards and maintain its temperature so that the coolant enters the reactor core at temperature T_1. In the simplified model, the velocity of the coolant in the downcomer is supposed to be equal to the velocity of the coolant in the core. Heat transfer from core to the reflector is by conduction and convection and also heat transfer from the tank to the pool (Q_{36}+Q_{56}) across the wall of the tank is again by conduction and convection mechanisms.The difference in density of the coolant as a result of changes in coolant temperature, acted upon by gravity, results in a cycle of coolant motion and the determination of the various coolant temperatures at different positions, as well as the coolant velocity, represents the real transient thermo-hydraulic analysis of the reactor.

Mathematical formulation of the model: The simplified model of the heat exchange process in the MNSR is obtained by an energy balance based on the average temperatures in the various components of the research reactor. This model, known as lumped parameter is suitable for this type of transients (Kazeminejad, 2008). For coolant flow pattern shown in Fig. 1, the partial differential equations for diffusion of heat within each fuel element in the reactor core region is written as:

$$c_{pf}\,\rho_f\,\frac{\partial T_f}{\partial t}(r,t)=\frac{1}{r}\frac{\partial}{\partial r}\left[k_{fm}r\frac{\partial T_f}{\partial r}(r,t)\right]+q_v(r,t) \quad (1)$$

and for cladding region, it is expressed in the form:

$$c_{pc}\rho_c\,\frac{\partial T_c}{\partial t}(r,t)=\frac{1}{r}\frac{\partial}{\partial r}\left[k_c r\frac{\partial T_c}{\partial r}(r,t)\right] \quad (2)$$

where, c_{Pf} is heat capacity for the fuel element, c_{Pc} is heat capacity for the cladding, k_{fm} is thermal conductivity of fuel meat, k_c is thermal conductivity of cladding, T_c is cladding surface temperature, T_f is fuel meat surface temperature, and q_v is volumetric rate of heat generation. From steady-state thermal-hydraulics analysis, it has been demonstrated that the temperature difference between the fuel meat at the centre and the outside of the cladding is 1ºC for nominal power of 30 kW (Shi Shuankai, 1990; Akaho and Dagadu, 1999). To avoid any thermal resistance between the fuel meat and the cladding, as a result of close attachment of the thin cladding with the fuel meat, without any gas gap, it is assumed that at any power level attained during transients, the temperature at the fuel and cladding regions will be almost the same (Quian, 1990; Akaho and Maaku, 2002). Therefore,

$$T_c(t) = T_f(t) \quad (3)$$

Thus, the simplified equation based on the lump parameter technique using Eq. (1) and (2) becomes:

$$\frac{dT_c}{dt}=\frac{P(T)}{m_f c_{Pf}+m_c c_{Pc}}-\frac{A_s h^*}{m_f c_{Pf}+m_c c_{Pc}}T_c(t)+T(t)+\frac{A_s h^*}{m_f c_{P_f}+m_c c_{Pc}}T_{av}(t) \quad (4)$$

where P is the reactor power, A_s is the heated clad surface area, m_f and m_c refers to mass of fuel and clad respectively. The average bulk water temperature in the core (T_{av}) is expressed as:

$$T_{av} = 1/2(T_2+T_1) \quad (5)$$

where T_2 is temperature at which coolant leaves the core and T_1 is temperature at which coolant enter the core.

The heat transfer coefficient varies with power and time and is determined from full scale simulation experiment. The correlation (Zhang, 1993; Akaho and Maaku, 2002) used to calculate the heat transfer coefficient is expressed in the form:

$$h^*(t) = k_f(t)/d_{eq}\, n(Gr_f(t).Pr_f(t))^m \quad (6)$$

The constant n and exponent m only valid for MNSR core are:

$$n = \begin{cases} 0.68 \\ 0.174 \end{cases} \quad m = \begin{cases} 1/4 \\ 1/3 \end{cases}$$

and
$Gr_fPr_f < 6\times10^6$ laminar flow
$Gr_fPr_f > 6\times10^6$ turbulent flow
d_{eq} is equivalent diameter and k_f is thermal conductivity of coolant.

From the assumption that there is no thermal resistance between fuel meat and cladding and that $T_c(t) \approx T_f(t)$ at any power level attained during the transients, the conservation of energy equation is expressed in the form:

$$c_{Pav}\rho_{av}\frac{\partial T_{av}(t)}{\partial t} + c_{Pav}\rho_{av}\frac{\partial T_{av}(t)}{\partial z} = \frac{A_s}{A_c}h^*\left(T_c(t) - T_{av}(t)\right) \quad (7)$$

and

$$\frac{dT_{av}(t)}{dt} = \frac{A_s h^*_{av}}{A_c H c_{Pav}\rho_{av}}T_c(t) - \frac{A_s h^*_{av}}{A_c H c_{Pav}\rho_{av}}T_{av}(t) - \frac{U_{av}(t)}{H}T_2(t) + \frac{U_{av}(t)}{H}T_1(t) \quad (8)$$

where A_c is net cross sectional area of the core, and u(t) is coolant velocity, H is the active height of the core.

The core inlet temperature at any time $T_1(t)$ and the outlet temperature $T_2(t)$ is determined using the correlation between reactor power, the inlet/outlet temperature of the core and the height of inlet orifice channelH_{in} (Shi Shuankai, 1990) as:

$$T_2(t)-T_1(t) = (5.725+147.6H_{in}^{-2.674})P(t)^{(0.59+0.0019T!(T))} \quad (9)$$

Equation (9) is valid for only MNSR core which is cooled under natural convection condition. Based on the thermal-hydraulic design consideration, the height of the core inlet orifice channel for the MNSR core is chosen in order to control the coolant flow within the core in order to achieve large temperature difference between inlet and outlet temperatures of the core. For the upper part of the reactor vessel (the tank), the heat transfer equation is written as:

$$m_3 c_{p3}\frac{dT_3(t)}{dt} = A_{up}c_{pav}u(t)\left(\rho_2 T_2(t) - \rho_3 T_3(t)\right) - Q_{36} \quad (10)$$

where m_3 is mass of water in the upper part of the reactor vessel, Q_{36} is heat transfer to the pool, T_3 is coolant temperature for the region and ρ is fluid density, A_{up} is cross sectional area of the coolant flowing to zone 3 of Fig. 1.
Substituting for Q_{36} in Eq. (10) and simplifying gives:

$$\frac{dT_3(t)}{dt} = \frac{A_{up}c_{pav}u(t)\rho_2}{m_3 c_{p3}}T_2(t) - \frac{\left(A_{up}c_{pav}u(t)\rho_3 + A_{pol}h^*_{pol}\right)}{m_3 c_{p3}}T_3(t) + \frac{A_{pol}h^*_{pol}}{m_3 c_{p3}}T_6(t) \quad (11)$$

For the annular downcomer region, the heat transfer equation is written as:

$$m_5 c_{pav}\frac{dT_5}{dt} = A_{dc}c_{pav}u(t)\left(\rho_3 T_3(t) - \rho_5 T_5(t)\right) - Q_{56} + Q_{25} \quad (12)$$

where m_5 is mass of coolant in the region, T_5 is temperature of coolant in the region, A_{dc} is cross sectional area of downcomer, Q_{56} is heat transferred from downcomer region to the pool, Q_{25} is heat transferred from annular beryllium reflector to the downcomer region. Experimental evidence (Zhang, 1993) under

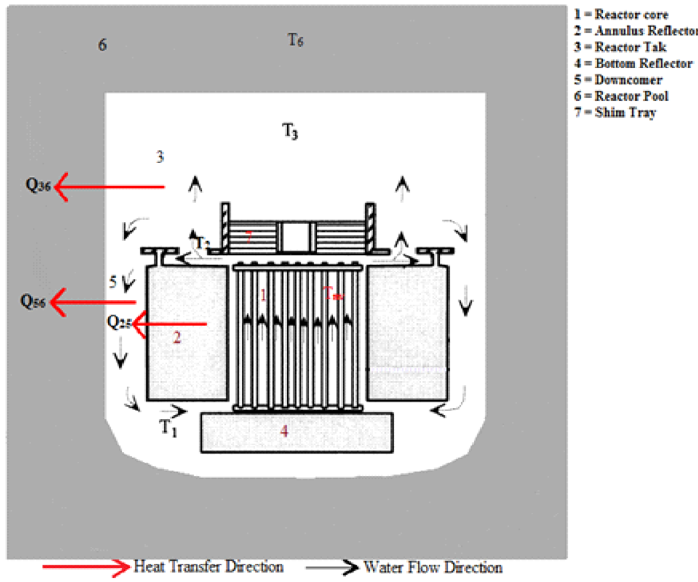

Fig. 1: A schematic diagram of the simplified model of GHAR-1

steady state condition and hydraulic feed-back experiments using prototype MNSR (Guo Chengzhan, 1990) showed that the temperature at the downcomer region is almost the same as the inlet temperature into the core. Hence $T_5(t) \approx T_1(t)$. Therefore,

$$m_5 c_{pav} \frac{dT_1(t)}{dt} = A_{dc} c_{pav} u(t)\big(\rho_3 T_3(t) - \rho_1 T_1(t)\big) - Q_{56} + Q_{25} \tag{13}$$

Substituting for Q_{56} and Q_{25} in Eq. (13) and simplifying, we obtain Eq. (14) as:

$$\frac{dT_1(t)}{dt} = \frac{A_{dc} u(t) \rho_3}{m_5} T_3(t) - \frac{\left(A_{dc} c_{pav} u(t) \rho_1 + A_{pol} h^*_{pol} + A_{dc} h^*_{dc}\right)}{m_5 c_{pav}} T_1(t) + \frac{A_{pol} h^*_{pol}}{m_5 c_{pav}} T_6(t) + \frac{A_{dc} h^*_{dc}}{m_5 c_{pav}} T_2(t) \tag{14}$$

Since the pool water has a large heat capacity, it is assumed that the heat transfer in the pool is isothermal to the ambient and the heat released from the core is totally absorbed by the water. The relevant heat transfer equation for the water pool region is simply written as:

$$M_6 c_{pav}\, dT_6(t)/dt = Q_{36} + Q_{56} \tag{15}$$

Further simplification of Eq. (15) after substituting for Q_{56} and Q_{36} gives:

$$\frac{dT_6(t)}{dt} = \frac{A_{pol} h^*_{pol}}{m_6 c_{pav}} T_3(t) + \frac{A_{pol} h^*_{pol}}{m_6 c_{pav}} T_1(t) - \frac{2 A_{pol} h^*_{pol}}{m_6 c_{pav}} T_6(t) \tag{16}$$

Since the core height is short we shall assumed that $\partial u/\partial z = 0$

The conservation of momentum equation for the reactor coolant water is written as:

$$-\frac{\partial P}{\partial z} = \rho_{av} \frac{\partial u(t)}{\partial t} + \rho_{av} u \frac{\partial u}{\partial z} + \frac{1}{2}\left[\frac{f_r}{d_{eq}} + \frac{\xi}{\Delta z}\right] \rho_{av} u^2(t) + \rho_{av} g \tag{17}$$

where f_r is skin friction, $f_r = 64/Re_d$ and ζ is drag coefficient dependent on flow regime, Re_d is Reynolds number.

If modifications to the actual cooling system are to be made, then the simple model described above will be too simple to solve the thermo-hydraulic problem of the reactor. Therefore the simple model has been modified to include the following;

The reactor vessel (tank) which is represented by Eq. (11) is divided into three sections, namely; upper section (ut), middle section (mt) and lower section (lt). The upper section extends above the upper ends of the fuel up to the upper edge of the tank. The middle section extends from the upper edge of bottom beryllium plate to the lower

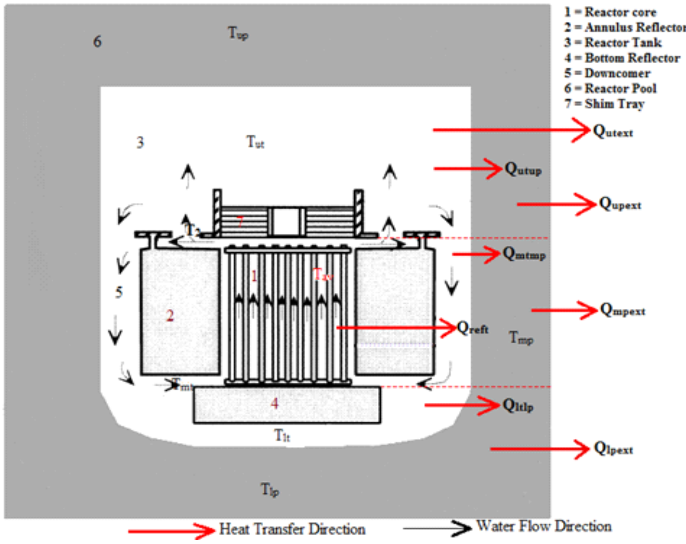

Fig. 2: A schematic diagram of the modified model of GHARR-1

level of the upper section (downcomer). The lower section extends from the lower level of the meddle section down to the end of the tank. It is important to note that only the upper section of the tank can be cooled (Fig. 2).

The reactor pool which is represented by Eq. (16) is also divided into three sections, namely; upper section (up), middle section (mp) and lower section (lp). The upper section extends above the upper ends of the fuel up to the upper edge of the pool. The middle section extends from the upper edge of the bottom beryllium plate to the lower level of the upper section. The lower section extends from the lower level of the middle section down to the end of the pool. Unlike the tank, the three parts can be cooled separately.

The velocities of the coolant in the core and in the downcomer region are no longer the same as it was in the simple model.

The modifications and assumptions when applied to the set of equations described under the simple model leads to the following set of equations for the modified model. The diffusion of heat within each fuel element in the reactor core region (Eq. 4) will now become:

$$\frac{dT_c}{dt} = \frac{P(t)}{m_f c_{pf} + m_c c_{pc}} - \frac{A_s h^*}{m_f c_{pf} + m_c c_{pc}} T_c(t) + \frac{A_s h^*}{m_f c_{pf} + m_c c_{pc}} T_{av}(t) - \frac{Q_{rad}}{m_f c_{pf} + m_c c_{pc}} \quad (18)$$

where the last term on the right hand side of Eq. (18) is the heat transferred from clad by radiation, expressed as $Q_{rad} = \sigma A_s^*(T_c^4(t) - T_{av}^4(t))$ and $\sigma = 4.96\times10^{-8}$ kcal/m^2 is a universal gas constant.

The conservation of energy equation within the fuel element in the reactor core region (Eq. 8) can be expressed as:

$$Hc_{pav}\rho_{av}\frac{dT_{av}(t)}{dt} = \frac{A_s}{A_c}h^*\left(T_c(t) - T_{av}(t)\right) - c_{pav}u_{av}(t)\rho_{av}\left(T_2(t) - T_1(t)\right) - Q_{reft} \quad (19)$$

where Q_{reft} is heat transferred through the reflector. Substituting for Q_{reft} in Eq. (19) and simplifying gives Eq. (20) below:

$$\frac{dT_{av}(t)}{dt} = \frac{A_s h^*_{av}}{A_c H c_{pav}\rho_{av}} T_c(t) - \frac{\left(A_s / A_c h^*_{av} + A_2 h_2^*\right)}{H c_{pav} p_{av}} T_{av}(t) - \left[\frac{u_{av}(t)}{H} - \frac{A_2 h_2^*}{H c_{pav}\rho_{av}}\right] T_2(t) \quad (20)$$

where

$$T_2 = 2T_{av} - T_{mt} \quad (21)$$

For the upper section of the tank, the heat transfer equation is written as:

$$m_{31}c_{put}\frac{dT_{ut}(t)}{dt} = A_c c_{put} u(t)\rho_{av}\left(T_2(t) - T_{mt}(t)\right) - A_{dc} c_{put}\rho_{ut} u_{dc}\left(T_{ut}(t) - T_{mt}(t)\right) \quad (22)$$

where Q_{utup} is heat transferred from the upper section of the tank to the upper section of the pool, Q_{utext} is heat transferred to an external heat exchanger (i.e., chiller) from the upper section of the tank, and u_{dc} is velocity of coolant in the downcomer, expressed as:

$$U_{dc} = \rho_2 u A_c / A_{dc}\rho_{mt} \quad (23)$$

Substituting for Q_{utup} and simplifying further Eq. (22) becomes:

$$\frac{dT_{ut}(t)}{dt} = \frac{A_c u(t)\rho_{av}}{m_{ut}} T_2(t) - \frac{\left(A_c u(t) p_{av} - A_{dc} p_{ut} u_{dc}\right)}{m_{ut}} T_{mt}(t) - \left[\frac{A_{dc}\rho_{ut}u_{dc}c_{put} + A_{pol}h^*_{pol}}{m_{ut}c_{put}}\right] T_{ut}(t) + \frac{A_{pol}h^*_{pol}}{m_{ut}c_{put}} T_{up}(t) - \frac{Q_{utext}}{m_{ut}c_{put}} \quad (24)$$

For the middle section of the tank, the heat transfer equation is written as:

$$m_{mt}c_{pmt}[dT_{mt}(t)/dt] = A_{dc}c_{pmt}u_{dc}\rho_{ut}(T_{ut}(t) - T_{mt}(t)) + Q_{reft} - Q_{utmt} - Q_{mtmp} \quad (25)$$

where Q_{utmt} is the heat transferred from the upper section of the tank to the middle section of the tank, Q_{mtmp} is heat transferred from the middle section of the tank to the middle section of the pool. Further simplification of Eq. (25) gives:

$$\frac{dT_{mt}(t)}{dt} = \frac{\left(A_{dc}c_{pmt}u_{dc}\rho_{ut} - A_{mt}h^*_{mt}\right)}{m_{mt}c_{pmt}} T_{ut}(t) + \frac{\left(A_{mt}h^*_{mt} - A_{dc}c_{pmt}\rho_{ut}u_{dc} - A_{mp}h^*_{mp}\right)}{m_{mt}c_{pmt}} T_{mt}(t) + \frac{A_2 h^*_2}{m_{mt}c_{pmt}} T_{av}(t) - \frac{A_2 h^*_2}{m_{mt}c_{pmt}} T_2(t) + \frac{A_{mp}h^*_{mp}}{m_{mt}c_{pmt}} T_{mp}(t) \quad (26)$$

For the lower section of the tank, the heat transfer equation is written as:

$$m_{lt}c_{plt}\,[d_{Tl}t(t)/dt] = Q_{utmt} - Q_{ltlp} \quad (27)$$

where Q_{ltlp} is the heat transferred from the lower section of the tank to the lower section of the pool.
Substituting for Q_{utmt}, Q_{ltlp} and simplifying, we obtain:

$$\frac{dT_{lt}(t)}{dt} = \frac{A_{mt}h^*_{mt}}{m_{lt}c_{plt}} T_{ut}(t) - \frac{A_{mt}h^*_{mt}}{m_{lt}c_{plt}} T_{mt}(t) - \frac{A_{lp}h^*_{lp}}{m_{lt}c_{plt}} T_{lt}(t) + \frac{A_{lp}h^*_{lp}}{m_{lt}c_{plt}} T_{lp}(t) \quad (28)$$

For the upper section of the pool, the heat transfer equation is written as:

$$m_{up}c_{pup}\,[dT_{up}(t)/dt] = Q_{utup} - Q_{upmp} - Q_{upext} \quad (29)$$

where Q_{upmp} is heat transferred from the upper section of the pool to the middle section of the pool, Q_{upext} is heat transferred to an external heat exchanger from the upper

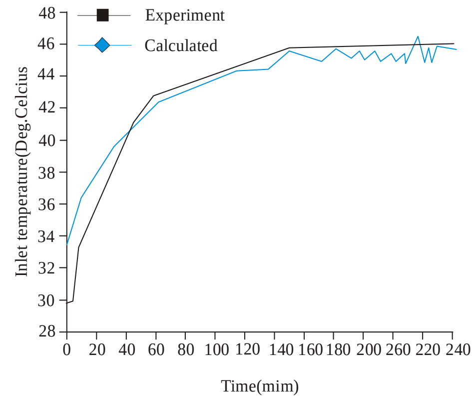

Fig. 3: Variation of core inlet temperatur

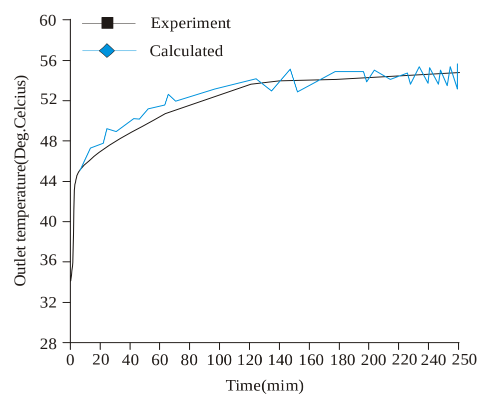

Fig. 4: Variation of core outlet temperature

section of the pool.Substituting for Q_{upmp}, Q_{utup} and simplifying, we get:

$$\frac{dT_{up}(t)}{dt} = \frac{A_{pol}h^*_{pol}}{m_{up}c_{pup}} T_{ut}(t) - \frac{2A_{pol}h^*_{pol}}{m_{up}c_{pup}} T_{up}(t) + \frac{A_{pol}h^*_{pol}}{m_{up}c_{pup}} T_{mp}(t) - \frac{Q_{upext}}{m_{up}c_{pup}} \quad (30)$$

For the middle section of the pool, the heat transfer equation is written as:

$$m_{mp}c_p\,[dT_{mp}(t)/dt] = Q_{mtmp} + Q_{upmp} - Q_{mplp} - Q_{mpext} \quad (31)$$

where Q_{mplp} is heat transferred from the middle section of the pool to the lower section of the pool, Q_{mpext} is heat transferred to an external heat exchanger from the middle section of the pool.

Substituting for the terms on the right hand side of Eq. (31) and simplifying, we obtain:

Table 1: Calculated reactor thermal-hydraulic parameters at $Q_{upext} = 0.0KW$ and $Q_{upext} = 30.0KW$

Parameter	Time (min)	Min. Value	Time (min)	Max. Value	Time (min)	Min Value	Time (min)	Max. Valu	Fina Value
T_{mt} (ºC)	0	29.60	240	45.975	0	29.60	240	44.224	44.224
T_{ut} (ºC)	0	29.60	240	46.185	0	29.60	240	44.08	44.08
T_{av} (ºC)	0	29.75	240	51.874	0	29.75	240	50.217	50.217
T_{up} (ºC)	0	29.98	240	32.126	240	28.758	1	29.98	28.758
T_c (ºC)	0	29.98	240	60.635	0	29.98	240	59.286	59.045
u (mm/s)	0	0.0	240	11.826	0	0.0	240	11.685	11.685
Cooling coil power (kW)		0.0					30.0		

Table 2: Calculated reactor maximum temperatures at different cooling coil powers.

	Max value	Final value	Max. value	Final value	Max. value	Final value
T_{mt} (ºC)	44.224	44.224	42.981	41.872	41.432	40.232
T_{ut} (ºC)	44.08	44.08	42.776	41.192	41.198	39.578
T_{av} (ºC)	50.217	50.217	48.789	47.748	47.336	46.303
T_{up} (ºC)	29.98	28.758	29.98	24.345	29.98	18.637
T_c (ºC)	59.286	59.286	58.248	57.408	57.221	56.384
u (mm/s)	11.685	11.685	11.581	11.503	11.501	11.140
Cooling coil power (kW)		30.0		60.0		120.0

$$\begin{aligned}\frac{dT_{mp}(t)}{dt} &= \frac{A_{mp}h^*_{mp}}{m_{mp}c_{pmp}}T_{mt}(t) - \frac{\left(A_{mp}h^*_{mp} + {}_2A_{pol}h^*_{pol}\right)}{m_{mp}c_{pmp}}T_{mp}(t) \\ &+ \frac{A_{pol}h^*_{pol}}{m_{mp}c_{pmp}}T_{up}(t) + \frac{A_{pol}h^*_{pol}}{m_{mp}c_{pmp}}T_{lp}(t) - \frac{Q_{mpext}}{m_{mp}c_{pmp}}\end{aligned} \quad (32)$$

For the lower section of the pool, the heat transfer equation is written as

$$m_{lp}c_{plp}\,[dT_{lp}/dt] = Q_{ltlp} + Q_{mplp} - Q_{lpext} \quad (33)$$

where Q_{lpext} is heat transferred to an external heat exchanger from the lower section of the pool.

Substituting for the right hand side of Eq. (33) and simplifying, we get:

$$\begin{aligned}\frac{dT_{ip}(t)}{dt} &= \frac{A_{lp}h^*_{lp}}{m_{lp}c_{plp}}T_{lp}(t) - \frac{\left(A_{lp}h^*_{lp} + A_{pol}h^*_{pol}\right)}{m_{lp}c_{plp}}T_{lp}(t) \\ &+ \frac{A_{pol}h^*_{pol}}{m_{lp}c_{plp}}T_{mp}(t) - \frac{Q_{ipext}}{m_{lp}c_{plp}}\end{aligned} \quad (34)$$

Equation (18), (20), (24), (26), (28), (30), (32) and (34) constitute the model sought in terms of lumped parameter. The above system of first order differential equations is solved numerically to provide the reactor transient response. The numerical solution was accomplished using Matlab/Simulink tool which is a commercial software package with the capability of modelling dynamical and control systems.

Variation of ten reactor thermal-hydraulic parameters with time are determined, these are: inlet temperature of the coolant, outlet temperature of the coolant, average coolant temperature in the core, coolant velocity, fuel temperature, temperature of reactor tank upper section, temperature of reactor tank lower section, temperature of pool upper section, temperature of pool middle section and temperature of pool lower section.

The following additional assumptions were made:

- The Power Establishment Time (PET) of GHARR-1 is not less than 2 min, during which the power rises linearly from zero to its pre-selected value.
- The temporal generation function of the reactor power is described by the function:

$$\begin{aligned}&P(t) = (RP/PET)\ t, \text{ for } 0 \le t \le PET \\ &P(t) = RP, \text{ for } t \ge PET\end{aligned} \quad (35)$$

RESULTS AND DISCUSSION

The equations describing the modified model used in this work were tested by comparing the results to the experimental data from GHARR-1. For the GHARR-1, the maximum license power is 30 kW but the normal operating power is 15 kW. The experimental results were obtained when the reactor was operated at 15 kW. The results show the time behaviour of the inlet and outlet temperatures of the coolant entering and leaving the reactor core as seen in Fig. 3 and 4, respectively.

Figure 3 and 4 illustrate the comparison between the calculated and experimental results for the case of no cooling applied to neither the pool water nor the tank water. This means $Q_{upext} = Q_{mpext} = Q_{1pext} = Q_{ufext} = 0$. The power establishment time is equal to 6 min from reactor start-up to the preset value.

It can be observed from the results of Fig. 3 and 4 that good agreement is found between the model predictions and the experimental results. The influence of the cooling coil power, which cools the upper section of the pool, on various temperatures in the reactor was analysed at different cooling coil powers.

Table 1 show the variation of reactor transient thermal-hydraulic parameters when the cooling coil power is 0.0 and 30.0 kW. It is observed from the table that all maximum values of the reactor thermal-hydraulic parameters decrease when the cooling coil power is increased from 0.0 to 30.0 kW. With the exception of the pool temperature which begins to decrease from the first minute of reactor operation, all the other reactor thermal-hydraulic parameters reaches the maximum value at the last minute of operation as a result of heat transfer from pool.

The effect of a decrease of reactor thermal-hydraulic parameters due to increase in cooling coil power is summarised in Table 2. It is observed that the difference between the no-cooling case (i.e., 0.0 kW cooling coil power) and the cooling coil power of 120 kW power case for the clad temperature is 4.3ºC and for the average coolant temperature in the core is 5.5ºC. It is further observed that the decrease of the maximum of the various temperatures is relatively small compared with the increase of the cooling coil power.

It should be noted that the effect of the cooling coil here is that the reactor begins to loose the steady thermo-hydraulic state when the cooling coil power is just 30 kW and the reactor will be always in transient thermo-hydraulic conditions. This means that higher cooling coil power will have some consequences on the stability of the neutronics of the reactor. Thus, to have a neutronically stable reactor, the cooling coil power should not be great. If this is the case, there will not be efficient cooling of the water passing through the core and so there will be no real advantage from cooling the water of the upper section of the pool.

CONCLUSION

A mathematical model is presented that permits to predict the reactor thermal-hydraulic behaviour of Ghana research reactor-1. The model is based on a lumped parameter description and is completely defined by Eq. (18), (20), (24), (26), (28), (30), (32) and (34). In the present work the model was applied to study the effect of the cooling coils of the pool upper section on the reactor thermal-hydraulic parameters by modifying the model described in (Zhang, 1993). Based on the results obtained, the following conclusions can be drawn. The maximum values of the reactor thermal-hydraulic parameters decrease when the cooling coil power is increased. The reactor begins to loose the steady thermo-hydraulic state when the cooling coil power is just 30 kW. To have a neutronically stable reactor, the cooling coil power should not be high as higher cooling coil power has consequences on the stability of the neutronics of the reactor. The model provides the understanding of the influence of the cooling coil power, which cools the upper section of the pool, on various temperatures in the reactor.

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